

NON-PUBLIC?: N
ACCESSION #: 9511070156
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Brunswick Steam Electric Plant, Unit 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000325

TITLE: Automatic Reactor Shutdown Due to Electro-Hydraulic
Control Pressure Regulator Malfunction
EVENT DATE: 07/13/95 LER #: 95-15-01 REPORT DATE: 11/01/95

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Steve F. Tabor, Regulatory
Affairs Specialist TELEPHONE: (910) 457-2178

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: TA COMPONENT: PC MANUFACTURER: G080
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 13, 1995, at 1158 hours, the Unit 1 reactor automatically shutdown (SCRAM) from 100% power when reactor pressure perturbations caused by a malfunctioning Electro-Hydraulic Control (EHC) pressure regulator created reactor power fluctuations. Average Power Range Monitors E and F generated a full Reactor Protection System (RPS) trip as a result of the fluctuating reactor power. The automatic isolation and/or actuation of the primary containment isolation system groups 1,2,3,6, and 8 occurred as designed. With the reactor stable and SCRAM signals reset, a full RPS actuation and the expected Engineered Safety Feature (ESF) actuations occurred on Unit 1 at 1433 hours due to a momentary shrink in reactor water level below the low level 1 setpoint. The EHC malfunction was localized to four pressure regulator "A" circuit boards. The cause of the component malfunction is still indeterminate following failure mode testing by General Electric Co. The second event

occurred following the cycling of a safety relief valve which was being used to control reactor pressure. Prior to Unit 1 startup with the reactor in hot shutdown, two additional full RPS logic actuations and expected ESF actuations occurred on July 14, 1995, at 2254 hours, and on July 15, 1995, at 0425 hours, when a momentary perturbation of the reactor water low level channels A2/B2 instrument sensing lines resulted in an invalid low level 1 trip signal. The momentary perturbations resulted from pressure spikes on the "B" reference leg sensing line, which occurred when actual reactor water level was being lowered to a point below the reactor pressure vessel reference leg nozzle.

END OF ABSTRACT

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TITLE

Automatic Reactor Shutdown Due to Electro-Hydraulic Control Pressure Regulator Malfunction

INITIAL CONDITIONS

on July 13, 1995, Unit 1 was operating at 100% power with the "B" Electro-Hydraulic Control (EHC) pressure regulator controlling reactor pressure. Due to drift problems, the "A" EHC pressure regulator had been biased low to prevent its control of the EHC system. The Emergency Core Cooling Systems were operable.

EVENT NARRATIVE

Between July 13, 1995, at 1158 hours, and July 15, 1995, at 0425 hours, Unit 1 experienced four events which are reportable in accordance with the requirements of 10 CFR 50.73. The following information provides a description of each of these events:

Event 1:

On July 13, 1995, at approximately 1150 hours, the "A" EHC pressure regulator began drifting erratically. At approximately 1153 hours, the "A" EHC pressure regulator signal drifted above the "B" EHC pressure regulator and assumed control of the EHC system. The malfunctioning pressure regulator resulted in a continuing decrease in reactor pressure by further opening the main turbine control valves and cycling the bypass valves (BPVs). The cycling BPV operation caused steam flow and pressure perturbations which resulted in reactor power fluctuations. At 1155

hours, Average Power Range Monitor upscale alarms were received. At 1158 hours, while Operations personnel were preparing to manually insert a reactor SCRAM, reactor power spiked due to bypass valve cycling. Consequently, APRMs E and F generated upscale trip signals (116% setpoint) which resulted in a full Reactor Protection System (RPS) trip and automatic reactor shutdown (SCRAM). All control rods inserted as required.

With EHC maintaining a lower reactor pressure, the Primary Containment Isolation System (PCIS) low pressure setpoint of 850 psig with the reactor mode switch in Run permissive for closure of the Main Steamline Isolation Valves (MSIVs) was satisfied when the reactor shutdown. Consequently, the MSIVs closed. The MSIV closure and reactor SCRAM resulted in a decrease in the reactor water level below the low level 1 and 2 setpoints (162.5" and 112", respectively). As designed, the low level 1 signal resulted in a PCIS Group 2 (Drywell Floor and Equipment Drains), Group 6 (Containment Atmospheric Control), and Group 8 (Shutdown Cooling) valve isolations. The low level 2 condition resulted in the initiation of the High Pressure Coolant Injection (HPCI) system, initiation and automatic injection of the Reactor Core Cooling Isolation (RCIC) system, Group 3 isolation (Reactor Water Cleanup), Reactor Building Ventilation system isolation, Standby Gas Treatment system initiation, and the trip of the reactor recirculation pumps. Reactor water level recovered before HPCI automatically injected.

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During verification of the automatic actuations that occurred during this event, the Division I Containment Hydrogen/Oxygen Monitor return isolation valve, 1-CAC-SV-1215E, exhibited dual position indication. Further verification determined that the affected return line had isolated due to the closure of the upstream valve. Upon discovering the dual indication condition, operations cycled the 1-CAC-SV-1215E valve to the closed position utilizing the valve control switch and observed a closed indication. The intermittent problem with the valve was investigated and corrective maintenance performed to ensure the operability of the valve prior to Unit 1 startup.

The HPCI, RCIC, and Control Rod Drive (CRD) systems were used to establish and maintain reactor vessel level. The safety relief valves (SRVs) were cycled and HPCI operated to control reactor pressure as necessary. At 1232 hours, the reactor SCRAM was reset.

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) in that an automatic actuation of ESF systems including the RPS occurred.

Following the SCRAM, closure of the MSIVs and the isolation of the RWCU system resulted in the isolation of the normal feed and condensate system from supplying feedwater to the vessel for level control. The SRVs and HPCI were used to control reactor pressure between 800 and 1000 psig. To provide level control and to makeup inventory reductions incurred during SRV pressure control, RCIC and HPCI were used to inject water from the Condensate Storage Tank (CST). CRD flow to the reactor vessel bottom head region was maintained until 1232 hours when the reactor SCRAM signal was reset. These actions placed additional cold water in the bottom head region of the vessel. At approximately 1245 hours, the RWCU system was restored to promote recirculation after the SCRAM signal was reset; however, having greater than a 145 degrees F differential temperature limitation between the reactor dome and bottom head area as defined by the Technical Specifications, the reactor recirculation pumps were not restarted.

The no-flow conditions combined with cold water injection caused the pressure-temperature limit curve as delineated in the Technical Specification Figure 3.4.G.1-1 to be exceeded. Additionally, reactor coolant stratification occurred in the reactor vessel bottom head region causing the localized cooldown in this area to exceed 100 degrees F/hour; however, the Technical Specification limit for vessel cooldown as delineated in Section 3.4.6.1.b was not exceeded.

In accordance with the action requirements of Technical Specification 3.4.6.1, an evaluation was performed to determine the effects of the above conditions on the fracture toughness properties of the reactor coolant system. The evaluation determined that the reactor coolant system remained acceptable for continued operation.

The cooldown event, although not reportable in accordance with the requirements of 10 CFR 50.73, has been included in this report to provide a complete description of the initial event and associated consequences.

Event 2:

At 1432 hours, with control rods fully inserted and Unit 1 in the hot shutdown condition, a full RPS actuation occurred due to a momentary decrease in reactor water level below the low level 1 setpoint. The decrease in reactor

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water level occurred following the closure of SRV A, which had been opened for approximately four minutes to control reactor pressure. The

reactor pressure band had previously been lowered to 600-1000 psig to reduce the reactor vessel dome and bottom head differential temperature to within 145 degrees F to support the restart of the reactor recirculation pumps. Prior to opening SRV A reactor water level was being maintained at approximately 208" to assist natural circulation and HPCI/RCIC were not injecting for level control due to reactor high water level trips. SRV A was closed with reactor pressure at approximately 620 psig and an indicated reactor water level of approximately 192". When SRV A closed, reactor water level momentarily decreased to approximately 155". As designed, the low level 1 signal resulted in PCIS Group 2 (Drywell Floor and Equipment Drains), Group 6 (Containment Atmospheric Control), and Group 8 (Shutdown Cooling) valve isolations. The ESF systems responded as designed. Following the momentary decrease, reactor water level returned to approximately 170" and HPCI/RCIC were initiated to restore level to greater than 210". The RPS trip signal was reset at 1442 hours. Use of the SRVs to control reactor pressure continued at this time without additional ESF actuations.

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) in that an automatic actuation of ESF systems including the RPS occurred.

Event 3:

On July 14, 1995, at 2254 hours, with the unit in the hot shutdown condition and reactor water level at approximately 219" to 217", Unit 1 received a spurious trip of the low reactor water level A2/B2 RPS logic channels, resulting in a reactor SCRAM. Additionally, PCIS Group 2 (Drywell Floor and Equipment Drains) outboard isolation, Group 6 (Containment Atmospheric Control) full isolation, and Group 8 outboard isolation (Shutdown Cooling) occurred. The ESF systems responded as designed. After confirming that actual reactor water level had remained constant during this event, the RPS trip signal was reset at 2305 hours. The group isolations were reset and associated equipment realigned by 2320 hours.

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) in that an automatic actuation of ESF systems including the RPS occurred.

Event 4:

On July 15, 1995, at 0425 hours, with the unit in the hot shutdown condition, the A loop of RHR operating in the shutdown cooling mode (to reduce the differential temperature between the reactor dome and bottom head to assist restart of the reactor recirculation pumps), and reactor

water level at approximately 217", Unit 1 experienced a second full RPS actuation due to a spurious trip of the reactor water low level A2/B2 RPS logic channels. The same actuation and isolations of the RPS and PCIS that occurred during the previous level perturbation event on July 14, 1995, resulted from the spurious trip signal. The ESF systems responded as designed. After confirming that actual reactor water level had remained constant during this event, the RPS trip signal was reset at 0428 hours. The isolations were reset, associated equipment realigned by 0445 hours.

This event is being reported in accordance with the requirements of 10 CFR 50.73

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(a)(2)(iv) in that an automatic actuation of ESF systems including the RPS occurred.

Following the Plant Nuclear Safety Committee review and approval of an investigation into the cause of these events and review of plant readiness for startup, Unit 1 startup commenced on July 15, 1995, and the main generator was synchronized to the electrical grid system on July 17, 1995, at 0901 hours.

CAUSE OF EVENTS

Event 1:

An incident investigation was initiated to determine the cause of the malfunctioning "A" EHC pressure regulator. A detailed troubleshooting plan was executed and a functional check of the EHC circuitry was performed. The cause of the malfunctioning EHC pressure regulator was localized to four circuit boards. Following failure mode analysis by General Electric Co., the root cause of the malfunction is still indeterminate.

Event 2:

This event was caused by a lack of appreciation for the magnitude of reactor water level change due to shrinkage which can result from a reactor pressure reduction when using an SRV to control reactor pressure at the lower range of a lower than normal pressure band. Neither training nor plant experience had prepared operations personnel to anticipate the magnitude of level change due to shrinkage that occurred. Typical simulator training scenarios establish pressure bands of 800-1000 psig with injection from at least one inventory makeup source available.

In this event makeup was inhibited due to a high level condition which was established to assist natural circulation. Additionally, a lower pressure band of 600-1000 psig had been established to assist in the restart of the reactor recirculation pumps. Based on past plant experience and training, the operators anticipated the closure of SRV A would result in a momentary shrink in reactor water level to approximately 170-175" from a starting point of 192". When SRV A was closed reactor water level decreased below the low level 1 setpoint (162.5") resulting in a full RPS actuation.

Following the event, several simulator exercises were conducted with conditions designed to duplicate as close as possible the plant conditions at the time of this event. During each of the exercises the simulated reactor level and pressure responded similarly to actual plant conditions experienced during this event (i.e., low level 1 setpoint was reached during pressure reduction).

Events 3 and 4:

These events were caused by spurious perturbation of the low reactor water level instrumentation sensing lines associated with the A2/B2 RPS logic channels. Initial troubleshooting into the cause of the spurious level perturbations discovered air in the variable line side of the affected instruments. Backfill of the affected instruments and their sensing lines and verification that the instruments were functioning properly were performed prior to startup.

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Further investigation into the cause of the spurious level perturbations determined that operating with reactor water level above the reference leg nozzle, while operating in the Hot Shutdown condition, generated pressure spikes in the reactor water level instrument condensing chambers. The pressure spikes were caused by the effects of steam entrainment and bubble collapse which occurred in the "B" reactor pressure vessel nozzle and reference leg. The spikes occurred due to increasing reactor water level above the nozzle at approximately 219" and then lowering the level to below the nozzle area. Control of reactor water level in this manner allowed water to exit the nozzle and condensing pots at the same time steam was re-entering the reference leg. This caused a momentary upward pressure spike in the reference leg, which resulted in a momentary indicated low reactor water level.

CORRECTIVE ACTIONS

Event 1:

The four electronic circuit boards comprising the "A" EHC pressure regulator circuitry were replaced, and the EHC system satisfactorily tested and returned to service before the restart of Unit 1 on July 15, 1995.

Event 2:

Following level recovery, directions were provided by shift supervision that when making pressure reductions using SRVs with level makeup reduced (e.g. HPCI/RCIC tripped on high level), limit the pressure reduction to 200 psig increments to assure level control following SRV closure.

Training of appropriate Operations personnel to ensure that large reactor pressure reductions should be limited based on vessel level considerations will be performed by October 31, 1995.

An evaluation to determine the need for initial licensed operator training and/or retraining on the effects of large reactor pressure reductions on vessel level will be performed by October 31, 1995.

Events 3 and 4:

As an interim measure, a review of this event including the effects of maintaining reactor water level near the reactor pressure vessel nozzle and the potential for reactor water level instrumentation reference leg perturbation was conducted with all shift superintendents.

Formal Operator training on the lessons learned from this event will be completed by January 15, 1996.

SAFETY ASSESSMENT

These events are of minimal safety significance in that the plant responded as designed and consistent with the analyses presented in the Updated Final Safety Analysis Report.

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In accordance with the requirements of the Technical Specification, an evaluation to determine the effects of exceeding pressure-temperature limits and the reactor coolant stratification that occurred during this event was performed. This evaluation concluded that the structural adequacy of the Unit 1 reactor vessel, based on fatigue and fracture toughness margins has been maintained. As a result, the reactor coolant

system was determined to be acceptable for continued operation.

PREVIOUS SIMILAR EVENTS

Similar events involving ESF actuations which resulted from instrument sensing line perturbations were reported in LERs 1-91-018 and 1-94-008.

Similar events involving ESF actuations which resulted from SRV pressure control operations and EHC pressure regulator malfunction were not identified.

EIIS COMPONENT IDENTIFICATION

System/Component EIIS Code

Main Turbine System TA

Reactor Protection System JD

Primary Containment Isolation System JM

Safety Relief Valve RV

High Pressure Coolant Injection BQ

Reactor Core Cooling Isolation BN

Containment Atmospheric Control IK

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Enclosure

List of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated

Committed

Commitment date or

outage

1. Training of appropriate Operations personnel to 10/31/95 ensure that large reactor pressure reductions should be limited based on vessel level considerations will be performed.

2. An evaluation to determine the need for initial 10/31/95

licensed operator training and/or retraining on the effects of large reactor pressure reductions on vessel level will be performed.

3. Formal Operator training on the lessons learned 1/15/96 from the reactor water level perturbations that occurred on July 14 and 15, 1995 will be performed.

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CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461-0429

SERIAL: BSEP-95-0577
10 CFR 50.73

NOV 01 1995

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325/LICENSE NO. DRP-71
SUPPLEMENTAL LICENSEE EVENT REPORT 1-95-015

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Supplemental Licensee Event Report. The original report fulfilled the requirement for a written report within thirty (30) days of a reportable occurrence and was submitted in accordance with the format set forth in NUREG-1022, September 1983.

Please refer any questions regarding this submittal to Mr. K. A. Harris at (910) 457-3312.

Sincerely,

W. Levis, Director-Site Operations
Brunswick Nuclear Plant

SFT/

Enclosures

1. Supplemental Licensee Event Report
2. Summary of Commitments

cc: Mr. S. D. Ebnetter, Regional Administrator, Region II
Mr. D. C. Trimble, Jr., NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

*** END OF DOCUMENT ***
